INTERNATIONAL ATOMIC ENERGY AGENCY

NUCLEAR DATA SERVICES

DOCUMENTATION SERIES OF THE IAEA NUCLEAR DATA SECTION

IAEA-NDS-141
Rev. 2, Oct. 1993

THE INTERNATIONAL REACTOR DOSIMETRY FILE

(IRDF-90 Version 2)

Assembled by

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Abstract: This document describes the contents of the new version of the International Reactor Dosimetry File IRDF-90 Ver. 2 which contains recommended neutron cross-section data to be used for reactor neutron dosimetry by foil activation. It also contains selected recommended values for radiation damage cross-sections and benchmark neutron spectra. This library supersedes all earlier versions of IRDF. It is available on magnetic tape or on a set of PC diskettes from the IAEA Nuclear Data Section, costfree, upon request.

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1. Introduction

Since the first release of the IRDF-90 v. 1 file in summer 1990 we have received many comments from its users. The main problems were identified in the covariance information (Files 33). Since then also some new evaluations appeared which were not available at the time of the release of version 1. Six new covariance files were added to the file. They were also not available before. In its present form the file contains 58 cross-sections of dosimetry reactions, all with complete covariance information. 9 new dosimetry reactions were added compared to version 1. The IRDF-90 version 2 contains 39 neutron dosimetry reaction cross-sections from the latest revisions of the ENDF/B-6 [1], 14 evaluations made by Prof. H Vonach and his co-workers at the IRK in Vienna [2] and 5 evaluations by the specialists from the Chinese Nuclear Data Center in Beijing, prepared specially for this file under contract with the IAEA [3]. The data in the original ENDF-6 format were processed to 640 group extended SANDII format in the Nuclear Data Section of the IAEA using the processing codes LINEAR, RECENT and GROUPIE by D.E. Cullen [4]. The covariance information is not processed by these codes and it is contained in IRDF-90 in the original ENDF-6 format.

2. Contents of the IRDF-90

The list of reactions and the origins of evaluations are given in Table 1. As we did not have any new sets of standard damage cross-sections or of standard and reference neutron spectra the ones from IRDF-85 were kept here with the same special notations. The damage cross-sections and neutron spectra are in the ENDF-5 format.

Data Content:

File 1	Cross section data in ENDF/B-VI format 25211 records for 58 reactions
File 2	Damage cross sections in ENDF/B-V format 754 records for 4 materials
File 3	Spectra data files in ENDF/B-V format 1598 records for 10 benchmark neutron fields

In File 3 neutron spectra for the following benchmark neutron fields are given

Cf-252	spontaneous fission - NBS Evaluation
U-235	thermal fission - NBS evaluation
U-235	thermal fission - ENDF/B-V evaluation
ISNF	Intermediate-energy standard neutron field
CFRMF	Coupled fast reactivity measurement facility
BIG-TEN	10% enriched uranium cylindrical critical assembly (LANL)
SIGMA-SIGMA	Coupled thermal/fast uranium and boron carbide spherical assembly (MOL)
ORR	Reactor in Oak Ridge National Laboratoy
YAYOI	Spectrum (JAERI)

Central zone flux of the NEACRP benchmark

All improvements in the file became possible only through efficient cooperation between Drs. H. Nolthenius, E. Zsolnay, and E. Szondi who were testing the file [5,6] and Drs. H. Vonach, S. Tagesen and D. Hetrick who made the necessary improvements in the covariance data files. Their contribution is gratefully acknowledged.

We would appreciate receiving any suggestions concerning further improvement of the quality of this file. Please send comments to:

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References

1.U.S. National Nuclear Data Center, Evaluated Nuclear Data File, ENDF/B-6, BNL, Upton, N.Y. (1990) and later revisions.

2.M. Wagner, H. Vonach. A. Pavlik, B. Strohmaier, S. Tagesen, J. Martinez-Rico, "Evaluation of Cross-Sections for 14 Important Neutron Dosimetry Reactions," Physics Data, 13-5, Karlsruhe, 1990.

3.C. Dunjiu, "Evaluations of Cross-Sections for Dosimetry Reactions," Final Report on Contract 5516, INDC(CPR)-024, 1991, Vienna.

4.D.E. Cullen, "The 1992 ENDF/B Preprocessing Codes", Report IAEA-NDS-39 Rev. 7, 1992.

5.E.M. Zsolnay, H. Nolthenius, "On the Quality of the Uncertainty Information in the International Dosimetry File IRDF-90," Report ECN-1-93-019, ECN, Petten, 1993.

6.H. Nolthenius, E.M. Zsolnay, E.J. Szondi, "Testing of the IRDF-90 Cross-Section Library in Benchmark Neutron Spectra," Reactor Dosimetry ASTM 1228, Harry Farrar IV, E. Parvin Lippincott, and John G. Williams, Eds., American Society for Testing and Materials, Philadelphia, to be published in 1994.

Table 1. Contents of the IRDF-90

E-6 = data taken over from ENDF/B-VI
Original = data evaluated for IRDF-90
Priv. Comm. = Private Communication

New evaluations introduced into the file are shown in **bold**.

Nuclide	IRDF MAT No.	Reactions and* Uncertainties	Author & Lab **	Date	Library of Origin
3-Li-6	325	3 105; 33 105	G. Hale et al., LANL	1989	E-6
5-B-10	525	3 1; 3 107; 33 107	G. Hale et al., LANL	1989	E-6
9-F-19	925	3 16; 33 16	M. Wagner et al., IRK	1991	Original
11-Na-23	**1123**	**3 102; 33 102**	**Yu Hanrong, CNDC**	**1990**	**Priv. Comm.**
12-Mg-24	1225	3 103; 33 103	M. Wagner Et al., IRK	1991	Original
13-Al-27	1325	3 103; 33 103	D. Hetrick, C.Y. Fu, ORNL	1990	Priv. Comm.
		3 107; 33 107	M. Wagner et al., IRK	1991	Original
15-P-31	1525	3 103; 33 103	M. Wagner et al., IRK	1991	Original
16-S-32	1625	3 103; 33 103	D. Hetrick, C.Y. Fu, ORNL	1991	Priv. Comm.
21-Sc-45	2126	2 151; **32 151**; 3 102; 33 102	**Z. Zhao, CNDC**	**1991**	**Priv. Comm**.
22-Ti-46	2225	3 103; 33 103	D. Hetrick, C.Y. Fu, ORNL	1989	Priv. Comm.
22-Ti-47	2228	**3 28; 33 28**; 3 103; 33 103	D. Hetrick, C.Y. Fu, ORNL	1990	E-6
22-Ti-48	2231	3 28; 33 28	C. Philis et al., ANL	1977	E-6
		3 103; 33 103	D. Hetrick, C.Y. Fu, ORNL	1990	Priv. Comm.
23-V-0	2300	3 107; 33 107	A. Smith, D. Smith, ANL	1990	Priv. Comm.
24-Cr-52	2431	3 16; 33 16	M. Wagner et al., IRK	1991	Original
25-Mn-55	2525	2 151; 3 16; 33 16; **3 102; 33 102**	K. Shibata et al., JAERI, ORNL	1988	E-6
26-Fe-54	2625	3 103; 33 103	D. Hetrick, et al., ORNL	1989	Priv. Comm.
26-Fe-56	2631	3 103; 33 103	C. Fu et al., ORNL	1991	E-6
26-Fe-58	2637	2 151; 3 102; **33 102**	N. Larson et al., ORNL	1989	E-6
27-Co-59	2725	3 16; 33 16	M. Wagner et al., IRK	1990	Original
		2 151; 3 102; **33 102** 3 107; 33 107	A. Smith et al., ANL	1990	E-6
28-Ni-58	2825	3 103; 33 103	N. Larson et al., ORNL	1989	E-6
		3 16; 33 16	M. Wagner et al., IRK	1990	Original
28-Ni-60	2831	3 103; 33 103	N. Larson et al., ORNL	1991	E-6
29-Cu-63	2925	3 16; 33 16	M. Wagner et al., IRK	1991	Original
		2 151; 3 102; **33 102** 3 107; 33 107	C. Fu et al., ORNL	1991	E-6
29-Cu-65	2931	3 16; 33 16	C. Fu et al., ORNL	1991	E-6
30-Zn-64	3025	3 103; 33 103	M. Wagner et al., IRK	1991	Original
39-Y-89	3925	3 16; 33 16	R. Howerton, A. Smith, D. Smith, LLNL, ANL	1991	E-6
40-Zr-90	4025	3 16; 33 16	M. Wagner et al., IRK	1991	Original
41-Nb-93	4125	3 16; 3 51; **3 102**	M. Wagner et al., IRK	1991	Original
		33 16; 33 51; **33 102**	**A. Smith et al., ANL, LLL**	**1991**	**E-6**
45-Rh-103	4525	3 51; 33 51	M. Wagner et al., IRK	1991	Original
47-Ag-109	**4731**	**3 102; 33 102**	**Z. Zhao, CNDC**	**1990**	**Priv. Comm.**
48-Cd-0	4800	3 1; 3 102	S. Pearlstein, BNL (translated from UK)	1991	E-690

Nuclide	IRDF MAT No.	Reactions and* Uncertainties	Author & Lab **	Date	Library of Origin
49-In-115	4931	2 151; **3 16**; **33 16**	**C. Dunjiu, CCNDC**	**1991**	**Priv. Comm.**
		3 51; 33 51 **3 102**; **33 102**	S. Chiba et al., ANL	1990	E-6
53-I-127	**5325**	**3 16**; **33 16**	**Z. Wenrong et al., CNDC**	**1991**	**Priv. Comm.**
64-Gd-0	6400	3 1; 3 102	Mixed from E-6 isotope data by N. Kocherov, IAEA	1990	Original
79-Au-197	7925	2 151; 3 102 33 102	P. Young, LANL	1989	E-6
		3 16; 33 16	M. Wagner et al., IRK	1991	Original
90-Th-232	9040	2 151; 3 18 3 102; 33 18 33 102	M. Bhat et al., BNL, ANL	1990	E-6
92-U-235	9228	2 151; 3 18 33 18	L. Weston et al., ORNL, LANL	1989	E-6
92-U-238	9237	2 151; 3 18 33 18; 3 102 33 102	L. Weston et al., ORNL, LANL	1989	E-6
93-Np-237	**9337**	2 151; 3 18; **33 18**	**F. Mann et al., HEDL, SRL**	**1978**	**E-4**
94-Pu-239	9437	2 151; 3 18 33 18	P. Young et al., LANL	1989	E-6
26-Fe-00	8000	ASTM Damage	Priv. Comm. W. Zijp Cross Sections	1979	Priv. Comm.
26-Fe-00	8001	Eur. Damage Cross Sections	Priv. Comm. W. Zijp	1979	Priv. Comm.
24-Cr-00	8002	Eur. Damage Cross Sections	W. Zijp, Petten	1985	Priv. Comm.
28-Ni-00	8003	Eur. Damage Cross Sections	W. Zijp, Petten	1985	Priv. Comm.

Note: * The following ENDF notations for reactions are used 1-total, 16-n,2n, 18-fission, 28-n,np, 102-nγ, 103-np, 107-na, 2 151 - resonance parameters. 51 means total population of the 1st level from all channels (not an ENDF notation); 3 - cross-section data file; 33 - covariance data file.

** The lab codes given under "Author & Lab" are as follows:

ANL	-	Argonne National Laboratory, Argonne Illinois
BNL	-	Brookhaven National Laboratory, Upton, N.Y.
CNDC	-	Chinese Nuclear Data Center
IAEA	-	International Atomic Energy Agency, Vienna
IRK	-	Inst. für Radiumforschung und Kernphysik, Vienna
JAERI	-	Japanese Atomic Energy Research Inst., Tokai
LANL	-	Los Alamos National Laboratory, New Mexico
LLNL	-	Lawrence Livermore National Laborarory, California
ORNL	-	Oak Ridge National Laboratory, Tennessee
Petten	-	Netherland's Energy Research Foundation, Petten
SRL	-	Savannah River Laboratory, South Carolina